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COMPARISON OF NOVovoronezh Unit 5 NPP AND SOUTH UKRAINE Unit 1 NPP Level 1 PRA Results

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ABSTRACT

This paper describes a study undertaken to explain the risk profile differences in the results of PRAs of two similar VVER-1000 nuclear power plants. The risk profile differences are particularly significant in the area of small steam/feedwater line breaks, small-small LOCAs, support system initiators and containment bypass initiators. A top level (limited depth) approach was used in which we studied design differences, major assumptions, data differences, and also compared the two PRA analyses on an element-by-element basis in order to discern the major causative factors for the risk profile differences. We conclude that the major risk profile differences are due to differences in assumptions and engineering judgment (possibly combined with some design and data differences) involved in treatment of uncertain physical phenomena (primarily sump plugging in LOCAs and turbine building steaming effects in secondary system breaks). Additional major differences are attributable to support system characteristics.

Keywords: PRA, risk profile, VVER-1000, comparison, FSU, sump plugging, turbine hall effect

NOMENCLATURE

AFW	auxiliary feedwater, backup to EFW
CCDP	conditional core damage probability
CDF	core damage frequency
DC	direct current
EFW	emergency feedwater
FASIV	fast acting steam isolation valve
HPI	high pressure injection
HPR	high pressure recirculation

HRA	human reliability analysis
HVAC	heating, ventilation, air conditioning
IE	initiating event(s)
ISLOCA	interfacing system LOCA
Level 3	PRA study of radiological consequences of accidents
LOCA	loss of coolant accident
LOOP	loss of offsite power
LPI	low pressure injection
LPR	low pressure recirculation
MFW	main feedwater
MLIV	main loop isolation valve
MSIV	main steam isolation valve
MWe	electric power output in megawatts
NPP	nuclear power plant
PRA	probabilistic risk assessment
PWR	pressurized water reactor
RCP	reactor coolant pump
RCS	reactor coolant system
RHR	residual heat removal
SG	steam generator
SW	service water
UPS	uninterruptible power supply
VVER	Soviet Union-designed pressurized light water reactor

1. INTRODUCTION

This paper describes a study [1] that was undertaken to explain differences in the calculated risk profile in the PRAs of two similar VVER type nuclear power plants. Both PRA studies [2,3] used similar methodology frameworks, employed the small event tree - large fault tree method and used the IRRAS/SAPHIRE and RELAP-5 computer codes for PRA and thermal-hydraulics

modeling, respectively. The motivation for the work described in this paper derives from the fact that application of the PRA methodology to VVER reactors is a relatively recent occurrence and is performed by institutions in the host country with oversight by Western experts and agencies. The goals of the two studies were twofold: to calculate the risk from these reactors (and suggest possible improvements), and to transfer the PRA methodology to the respective countries. Therefore, it is incumbent to try to understand the reasons for significant differences in the results of the two different PRAs that studied seemingly very similar plants in two different countries. The two PRAs were funded and managed by different organizations and were performed by different teams.

As a reference, our study used information developed at the IAEA workshop on "Comparison of PSA Level 1 for VVER-1000 Reactors," held in Erlangen, Germany in 1999 [4]. This meeting compared PRA results, methodologies and plant features for six VVER-1000 plants, among them NVNPP-5 and SUNPP-1. The results of this meeting provided a useful starting point and direction for some aspects of our research.

Both plants are of the VVER-1000 type (i.e., producing 1,000 MWe). However, neither is the "mainline" VVER-1000 (i.e., the VVER-1000-320), which was introduced later. Novovoronezh Nuclear Power Plant Unit 5 (NVNPP-5), located in Russia, is a VVER-1000-187, while South Ukraine NPP Unit 1 (SUNPP-1) is a VVER-1000-302. These are pressurized water type reactor plants, designed and built in the former Soviet Union and associated countries. There are a number of similarities between the two VVER-1000s studied in the two PRAs, while there are also some important design differences, and these should influence the results to a certain extent.

Tables 1 and 2 show the important differences in the calculated risk profiles. The major differences that are apparent from the tables are those related to steam/feedwater line breaks, LOCAs and transient event group contributors. While containment bypass initiators do not play a major CDF role in either study, the major difference in the magnitude of the CDF contribution in the two studies is notable (and this may also have a significant impact on Level 3 (i.e., population dose, considerations).

Table 1 - Major CDF Contributors, NVNPP-5

Category	NVNPP-5 CDF (/yr)	NVNPP-5 CDF (%)
Steam/feed line breaks FW header, small Steam outside, small Steam inside, small	2.7e-04	39
LOCAs Large break Pressurizer SV open Medium, unisolable	1.9e-04	27
Transients Blackout, non-LOOP	1.4e-04	20
Loss of Offsite Power	5.5e-05	8
Primary to Secondary Leaks	2.1e-05	3
Containment bypass	1.9e-05	3
TOTAL CDF	6.9e-04	100

Table 2 - Major CDF Contributors, SUNPP-1

Category	SUNPP-1 CDF (/yr)	SUNPP-1 CDF (%)
LOCAs Small-small Large	8.1e-05	54
Transients Scram actuation	2.5e-05	17
Primary to Secondary Leaks	2.5e-05	17
Loss of Offsite Power	1.6e-05	10
Steam/feed line breaks FW header, large Steam line, large	3.3e-06	2
Containment bypass	5.1e-09	0
TOTAL CDF	1.5e-04	100

Table 3 presents the risk profile in more detail, along with associated information on initiating event frequencies and conditional core damage probabilities (CCDPs). (Only the most significant contributing initiating events are included within each category, such that the frequencies of contributing initiating events within a group do not add up to the group frequency). Note that absolute CDF contributions are generally significantly higher at NVNPP-5 than at SUNPP-1, as are many important events' conditional core damage probabilities and IE frequencies. It can be seen that within transients, the risk contributions are significantly different, with support systems failures playing a commanding role at NVNPP-5,

Table 3 - Comparison of Initiating events grouping and frequencies

Initiating Event Group	NVNPP-5 IE Frequency (/yr)	NVNPP-5 CCDP	NVNPP-5 CDF (/yr)	SUNPP-1 IE Frequency (/yr)	SUNPP-1 CCDP	SUNPP-1 CDF (/yr)
Transients	4.3e+00	3.3e-05	1.4e-04	1.4e+00	1.8e-05	2.6e-05
Non-LOOP blackout: loss of dc, loss of sw, loss of switchgear room cooling in summer	1.0e-04	1.0e+00	1.0e-04	NM		
Inadvertent Opening all SG safety valves	1.5e-05	1.0e+00	1.5e-05	NM (spurious opening of one SGSV included under unisolable steamline breaks)		
Reactor scram SU	NA (different events included)			1.3e+00	9.0e-06	1.2e-05
Complete loss of main feedwater due to main feedwater discharge lines ruptures	In NVNPP-5, this is modeled under steamline/ feedwater line breaks			5.0e-03	8.4e-04	4.2e-06
Steam Line/Feedwater Line Breaks	7.0e-02	3.9e-03	2.7e-04	1.3e-02	2.5e-04	3.3e-06
			Small breaks 95% of CDF	Large breaks only CDF contributor		
Containment Bypass	1.6e-03	1.2e-02	1.9e-05	3.6e-07	1.4e-02	5.1e-09
Leakage outside containment in isolable part (CVCS system)	1.6e-03	1.2e-02	1.9e-05	3.6e-07	1.4e-02	5.1e-09
Leakage outside containment in unisolable part	4.4e-07	1.0e+00	4.4e-07	NS		
Loss of Offsite Power	3.3e-02	1.7e-03	5.6e-05	1.0e-02	1.6e-03	1.6e-05
Primary to Secondary Leaks	5.9e-03	2.9e-03	1.7e-05	4.8e-02	5.2e-04	2.5e-05
LOCAs inside containment	3.0e-02	6.3e-03	1.9e-04	2.5e-01	3.3e-04	8.2e-05
Large LOCA	1.0e-04	1.0e+00	1.0e-04	3.0e-04	6.6e-02	2.0e-05
Medium LOCA in unisolable part	1.0e-03	3.2e-02	3.2e-05	1.3e-04	4.2e-03	5.2e-07
Small-small LOCA	NS	NS	NS	2.4e-01	2.2e-04	5.4e-05

NS – the relevant numbers not shown in the report

NM – not modeled or screened out

NA – not applicable due to differences in design or modeling

while reactor scram and feedwater line rupture are dominant at SUNPP-1. The steam/feedwater line breaks, in particular the small breaks, play a dominant role in NVNPP-5, while secondary breaks generally play a relatively minor role in SUNPP-1. Moreover, the small secondary breaks do not appear at all in the results of the SUNPP-1 PRA (except, perhaps as transient contributors of no major significance). LOCAs are significant in both PRA study results, but the small-small primary system breaks are the dominant LOCA in SUNPP-1 PRA, while they do not appear at all in the NVNPP-5 PRA as contributors. The large break is the dominant LOCA in NVNPP-5, and the CCDP for large and medium LOCAs is much higher in NVNPP-5, as is the medium LOCA IE frequency. Support system failure transients are significant initiator contributors to CDF in the case of NVNPP-5, and do not appear as contributors in SUNPP-1. Several initiators are given a CCDP of 1 (guaranteed core melt) at NVNPP-5, while they either do not appear at SUNPP-1, or are have much smaller CCDPs.

The methodology employed in this analysis, to explain the calculated risk profile differences, was both inductive and deductive. The authors only had access to the two summary reports in English, which imposed limitations on depths of our analyses. Therefore, this was a top level approach to discern the major causative factors for major calculated risk profile differences.

The organization of the paper is as follows. In this section, we present the problem and the motivation for solving it. In Section 2, the methods used in the analysis are presented. Section 3 presents the results (i.e., what we think are the major reasons for the differences in the calculated risk profiles and why), while Section 4 presents the uncertainty issues, i.e., what additional information would be needed to pinpoint causes more precisely and to look at less significant calculated risk profile differences. Finally, the conclusions are given in Section 5.

2. METHOD OF ANALYSIS

Our study, which was limited in resources, used several methods to quickly discern the possible contributors to differences in the calculated risk profiles. One method was to break down the possible contributors into the areas of plant design, PRA assumptions and data differences, and to recast the PRA calculated risk profile differences in terms of possible causative factors from those three areas. In other words, this was a mostly deductive approach, in which we sought to see if

major calculated risk profile differences could be explained in terms of one or the other of the above possible causes. Another approach was to compare the two plants side by side in terms of major risk-important design features, and see if any differences encountered could translate into differences in the calculated risk profiles. This was a mostly inductive approach. The third method was to do an element-by-element comparison analysis of PRA components (e.g., initiating events, accident sequence logic, and so on) to see if any peculiarities or differences in the analysis of each individual element could translate into any of the observed calculated risk profile differences.

2.1 Design Features Comparison

As mentioned earlier, VVER reactors have many design and safety features comparable to those of Western PWRs. Some notable differences are: the use of horizontal steam generators, primary pressure is somewhat higher and primary temperature lower, the existence of main loop isolation valves on the primary side (isolate some LOCAs), the existence of fast acting steam isolation valves (FASIV) on the secondary side (isolate some secondary breaks, these valves are in addition to the MSIVs), the use of steam as motive force for some important valves, the lack of feed and bleed capability, the lack of extended blackout capability, the lack of HPR, vulnerability to steaming effects ("turbine hall effect"), the use of sturdy RCP seals, the use, generally, of three trains of safety equipment, and the use of event based operator procedures.

As far as the differences between the two VVERs in question are concerned, notable differences in gross design features are: the location of FASIV in NVNPP-5 upstream of the atmospheric steam dump valves (which complicates heat removal in many accidents), EFV pumps are not located in separate compartments in NVNPP-5 (possibly making them more vulnerable to the turbine hall effect), and the RCPs are manually tripped on loss of pump cooling in NVNPP-5 (vs. automatically at SUNPP-1).

As far as the differences in the frontline systems are concerned, Table 4 presents those. Notable is a common LPI line between trains, in the case of NVNPP-5, which introduces a common cause failure mode, higher LPI pump shutoff head at SUNPP-1 (LOCA concern), existence of operator procedure to counteract the effects of sump plugging at SUNPP-1 (LOCA concern), manual switchover of HPI suction from a smaller tank to a bigger tank at NVNPP-5, possibility of

using steam generator safety valves at NVNPP-5 for decay heat removal at low pressures (>10 atm), EFW connection to steam generators via MFW lines at NVNPP-5 (effect in certain secondary breaks), existence of a separate two-train AFW system at SUNPP-1 (in addition to the three train EFW system), and the possibility of employing the turbine driven MFW pumps at NVNPP-5, as there is steam cross-connect to Units 3 and 4 (VVER-440 units).

Table 5 presents the differences in support systems between the two plants. The most important differences are in switchgear room cooling requirements, DC power and service water dependencies, all in favor of SUNPP-1; any of these support system failures lead to core damage at NVNPP-5 with a high conditional probability. In addition, NVNPP-5 has a common cause failure mechanism between secondary cooling and RHR cooling, as both depend on a single emergency ac bus.

The most important observable design differences found in this section are those related to support systems (switchgear room HVAC, DC and SW), with some importance also attached to lack of compartmentalization of EFW pumps in NVNPP-5, the existence of operator procedures to countermand sump plugging during recirculation at SUNPP-1, common mode failure mechanisms involving the LPI/RHR and secondary cooling systems at NVNPP-5, and more favorable FASIV location at SUNPP-1.

2.2 Top Level Analysis

In this section, we ask the question: given the major observable calculated risk profile differences, can they best be explained in terms of differences in assumptions/engineering judgment, data employed or the design?

The answer is that in some cases, it is a mixture of the above causes, or the dominant cause cannot be ascertained due to limitations in documentation available to us.

The differences in LOCA contributions are due, in large part, to the sump plugging probabilities assigned in the two studies: the ones used in the NVNPP-5 PRA are much higher, as seen from Table 6. In large and medium LOCAs, sump plugging probability differences are directly responsible for the significant differences in the CCDP and the CDF. These differences in the sump plugging probability could be due to design (related to the sump screen and the thermal insulation used), the operator procedures (which are part of the "design"

category in this paper) or simply the fact that more conservative engineering judgment was used in the case of the NVNPP-5 PRA (where a large LOCA leads directly to core damage as a consequence). Note that these LOCA considerations are countermanded somewhat by the fact that SUNPP-1 LPI pumps have higher shutoff heads, which may help extend large LOCAs into smaller break sizes (as is observed in different LOCA category vs. size definitions at the two plants), thus potentially contributing to higher large LOCA frequency at that plant (both absolutely and relative to medium LOCA frequency). For example, at SUNPP-1, large LOCA is defined as having break sizes larger than 70 mm, whereas at NVNPP-5, large LOCA sizes are greater than 160 mm.

The differences in the secondary break contributions are due to assumptions. The SUNPP-1 PRA project does not consider steaming effects ("turbine hall effect") in this study, but such considerations are postponed for the next Phase of this study. The NVNPP-5 project does consider such effects, but incompletely, again leaving a more detailed treatment for the next Phase of that study. It also does a limited sensitivity study in which such effects guarantee core damage.

The differences in the support system initiators (part of transients category) contribution are probably due to differences in the design between the two plants.

The differences in the containment bypass contributors may be due to design and procedures related to the interface systems.

Note that limitations in the documentation prevent us from making more definitive statements.

2.3 Element-by-Element Analysis

In this approach, we look at each individual PRA element to discern if differences in the treatment of that element lead to calculated risk profile differences.

Initiating events. There are several initiating event categories that are considered in one of the PRAs but not in the other. Most are not significant, except for the common cause spurious relief valve opening (all SG safety valves or all atmospheric steam dump valves), common cause failures of support systems and small secondary breaks. All of these are treated in the NVNPP-5 PRA but not in the SUNPP-1 PRA, and all of them have a relatively high conditional core damage probability. In addition, the interfacing LOCA pathways in the two plants are not identical, and NVNPP-5

Table 4 - Frontline Safety System Differences

Frontline system	Design Parameters, NVNPP-5	Design Parameters, SUNPP-1
LPI System	Pumps: P<15 atm Same LPI train can be used in injection and cooldown mode by periodic manual switching between the modes; Lack of redundancy in planned cooldown mode as common line connects the three LPI trains to the RCS	Pumps: P<24b Separate LPI trains must be used for injection and planned cooldown modes
HPI and Makeup System	Suction from boric acid tank LPI tanks can also be manually aligned to be HPI suction source	LPI tanks provide suction to boric acid (high pressure makeup) pumps
LPR System		Operator procedures exist to counteract effects of sump screen Plugging (periodic change in running LPR train)
SG Safety Valves	Can be used for heat removal at low pressure (>10 atm)	
EFW	Connected to SGs via MFW lines Pumps: Q=65m ³ /h P=55 atm	Connected to SG via separate EFW dedicated lines Pumps: Q=150m ³ /h
AFW	NA	Connected to SGs via MFW lines Pumps: Q=150m ³ /h P=85 b
MFW	2 trains, steam driven, can be used post trip via steam cross-connect from Units 3 and 4	Not used post trip

Table 5 - Support System Differences

Differences
<p>UPS cooling required in SUNPP-1 > 8hrs in the summer; such cooling depends on offsite power</p> <p>In NVNPP-5, switchgear room cooling required in hottest period in the summer; loss leads to blackout and core damage</p> <p>In NVNPP-5, DC power loss leads to blackout and core damage (subject to operator recovery)</p> <p>In NVNPP-5, SW loss leads to blackout and core damage</p> <p>Common cause dependencies between secondary cooling and RHR cooling in NVNPP-5 via ac bus (not discussed in SUNPP-1 PRA)</p> <p>Differences related to system sharing and cross-connect with other units on site:</p> <p>Novo: steam cross-connect from Units 3&4; possible emergency power cross-connect from Unit 4; some SW source sharing with Units 3&4</p> <p>SUNPP: emergency power cross-connect with Unit 2 "possible", but not proceduralized; some normal power cross-connect with Unit 2; sharing of compressed air, NTV, non-essential SW and demineralized water with Unit 2 and possibility of limited demineralized water cross-connect to Unit 3</p>

considers additional pathways (HPI and LPI injection lines), which are unisolable (CCDP of 1) and which are not discussed in the SUNPP-1 PRA. Even for the isolable pathway, the NVNPP-5 initiating event frequency and the CDF are several orders of magnitude higher than the ones in the SUNPP-1 study, due to unknown reasons.

It is possible that the spurious relief valve opening is subsumed under the steam line break in the SUNPP-1 PRA. These are not important in that PRA, because only the recriticality concerns are treated (the environmental effects are left for the next phase of the project). The support system failures are either not initiators at SUNPP-1, or are probabilistically screened out; in any case the documentation is very limited. The small secondary breaks, due to non-treatment of the steaming effects in this phase, and due to lack of recriticality concerns, would fall into the category "administrative shutdown" at SUNPP-1. Such shutdowns are explicitly not considered in that study (though they are considered in the NVNPP-5 study).

Event trees. The NVNPP-5 event trees consider the unfavorable FASIV location (upstream of the atmospheric, and other, steam dump valves) and operator actions to prevent closure, as it takes an hour to reopen them, as well as other considerations related to this fact. These considerations tend to increase the NVNPP-5 conditional core damage probability in many initiators, as the FASIVs close on decreasing steam pressure. Small-small LOCA event trees involve the "UNSH" end state in large fraction of cases in the NVNPP-5 PRA. This is the undeveloped end state which will be analyzed in the shutdown study for the plant, as the evolution of the accident takes more than 24 hours' mission time (in other words there will be no core damage within that time period). As small-small LOCAs are significant contributors to the SUNPP-1 CDF, this is the reason for this category being different in the calculated risk profile (it is one of the dominant contributors in the SUNPP-1 risk profile, while it does not appear in the NVNPP-5 risk profile). It is not clear if different assumptions/criteria were used in the two studies, or there are some design differences, which translate into different timing of the event. The SUNPP-1 event trees give much more weight to recriticality concerns (e.g., in case of secondary breaks), and this again could be due to design differences or a more conservative approach. The NVNPP-5 event trees treat the steaming effects in secondary breaks, to a certain degree, while such treatment is absent from the SUNPP-1 event trees.

Systems. The NVNPP-5 PRA system unavailabilities are generally significantly higher than the ones in the SUNPP-1 study. This could be due to underlying data and translates to higher conditional core damage probabilities for NVNPP-5.

Data. There is a lack of complete information in the SUNPP-1 project Summary Report, but for the few components available (e.g., EFW pumps), the NVNPP-5 failure rates are significantly higher. The two studies rely on different generic failure data. The NVNPP-5 generic data comes from IAEA-TECDOC-478 [5] documents which compile VVER data from the field, while the SUNPP-1 project uses Western data (the T-book), noting the difficulties with the VVER data; Bayesian method is used to update this data with plant specific data.

HRA. The documentation was such that no meaningful comparison could be made. It is believed that this element probably does not account for the first order calculated risk profile differences.

3. RESULTS OF ANALYSIS

As a result of the analysis of the differences discussed above, it is concluded that the major differences observed and explained in Tables 7 and 8 are attributable to a combination of (1) differences in several initiating event groups that are included in one PRA and not the other, and (2) differences in modeling and assumptions regarding physical phenomena which lead to differences in conditional core damage probability associated with those phenomena. Some of these differences may be due to specific design differences, although this is not clear from the summary reports. More detailed descriptions of the studies are needed to determine further the relationship between design differences and the PRA treatment of the differences.

The major reasons for the observed differences, presented in Tables 7 and 8 are the following:

(1) Small feedwater and steamline breaks make significant contributions to the CDF in the NVNPP-5 PRA results, while results for these breaks are not presented in the SUNPP-1 PRA. Consideration of such initiators is deferred for Phase II of the SUNPP-1 study (due to complicated nature of turbine hall steam and flooding effects).

Table 6 - Sump Plugging Probabilities, NVNPP-5 PRA vs. SUNPP-1 PRA

LOCA	NVNPP-5 PRA Sump Plugging Probability	SUNPP-1 PRA Sump Plugging Probability
Large	1.0e+00	1.0e-02
Medium	3.2e-02	1.0e-05

Table 7 - Reasons for Risk Profile, NVNPP-5

Category	NVNPP-5 CDF (/yr)	NVNPP-5 CDF (%)	Comments
Steam/feed line breaks FW header, small Steam outside, small Steam inside, small	2.7e-04	39	Incomplete (in some aspects conservative) treatment of turbine hall steaming effects
LOCAs Large break Pressurizer SV open Medium, unisolable	1.9e-04	27	Small-small breaks -- unresolved end state left for next phase; high probability assigned to sump plugging
Transients Blackout, non-LOOP	1.4e-04	20	Support system initiators lead to blackout
Loss of Offsite Power	5.5e-05	8	
Primary to Secondary Leaks	2.1e-05	3	
Containment bypass	1.9e-05	3	A number of ISLOCAs included, including makeup line and high/low pressure injection lines
TOTAL CDF	6.9e-04	100	

Table 8 - Reasons for Risk Profile, SUNPP-1

Category	SUNPP-1 CDF (/yr)	SUNPP-1 CDF (%)	Comments
LOCAs Small-small Large	8.1e-05	53.9	Small-small breaks high frequency; large breaks sump plugging less conservative
Transients Scram actuation	2.5e-05	16.9	No blackout from support system initiators
Primary to Secondary Leaks	2.5e-05	16.5	
Loss of Offsite Power	1.6e-05	10.4	
Steam/feed line breaks FW header, large Steam line, large	3.3e-06	2.2	Small breaks, turbine hall steaming and flooding effects left for next phase
Containment bypass	5.1e-09	~0	Low frequency makeup line break; no other analyzed
TOTAL CDF	1.5e-04	100	

The steam/feedwater line breaks in NVNPP-5 contributes significantly to the core damage frequency, while these breaks are not treated at all in the SUNPP-1 PRA. In the case of SUNPP-1 the effect of steam in the turbine hall in the case of secondary breaks (the turbine hall effect) is specifically not treated in SUNPP-1, while a simplified treatment is employed in NVNPP-5. Both SUNPP-1 and NVNPP-5 postpone complete treatment of such effects to Phase II of their respective studies. The secondary breaks are treated and grouped at SUNPP-1 from the standpoint of effect on recriticality, which, in the absence of the turbine hall issue, is the major concern in this initiator group. Neither plant considered flooding in the present studies.

(2) LOCAs are significant contributors in both PRAs. While the small-small LOCA (diameter < 14 mm) is the major LOCA contributor in the case of SUNPP-1, results for this initiator are not presented in the NVNPP-5 summary report. The large break LOCA is dominant in the NVNPP-5 PRA. Large, medium and small LOCA CDF contributions (in the absolute sense) are much higher in NVNPP-5 PRA than in the SUNPP-1 study.

The contribution of LOCAs to the CDF in both the NVNPP-5 and SUNPP-1 PRAs are of similar magnitude (within about a factor of 2). However, there are significant differences in the CDF contributions from different LOCA break sizes. There are two major identifiable reasons for this:

- (a) The small-small LOCA end state in NVNPP-5 involving failure of RHR cooling after depressurization is not treated as a core damage end state, but instead is treated as "UNSH" (undeveloped and deferred for shutdown analysis). This was justified by NVNPP-5 on the basis that no core damage resulted in these sequences during the mission time. As a result, the small-small LOCA is a small contributor to the core damage frequency estimate for NVNPP-5. At SUNPP-1, all these sequences are treated as core damage sequences, resulting in a relatively large contribution of small-small LOCA to the CDF.
- (b) The large break LOCA (and to a certain extent medium and small LOCA too) is dominant in NVNPP-5 and is secondary in SUNPP-1 because of assumptions made regarding the sump plugging probability [in addition to the reason given in (a) above]. In NVNPP-5, the large break LOCA is effectively treated as an unprotected accident, whereas in SUNPP-1 the sump plugging probability

is assigned a relatively low probability, leading to a relatively low conditional core damage probability. Similarly, there are large differences in the sump plugging probabilities used for the medium (and probably small) LOCAs. There appears to be some justification for smaller sump plugging probabilities in the case of SUNPP-1, as a result of operating procedures that allow the operator to identify a pump whose line is plugged, to shut the affected pump and to switch to an unaffected one (which had been switched off early in the accident). It is not clear, however, if the difference in quantitative treatments is entirely justifiable on physical grounds. Additional reasoning in both PRAs would be helpful for an improved resolution of the differences.

(3) Several support system initiators in the case of NVNPP-5 lead to station blackout. Station blackout resulting from initiators other than loss of offsite power [non-LOOP blackout] is a significant contributor to the CDF in the case of NVNPP-5, whereas this type of initiator does not arise in the SUNPP-1 PRA. The contributors to non-LOOP blackout in NVNPP-5 are: loss of DC power, loss of service water and loss of switchgear room cooling in summer.

The transient initiating event category in NVNPP-5 is dominated by the non-loss of offsite power (non-LOOP) blackout, resulting from loss of DC power, loss of service water or loss of switchgear room cooling in summer. The SUNPP-1 PRA considers loss of a single DC bus, loss of one train of service water and loss of air conditioning. These failures are not treated as initiator groups because they do not lead to reactor trip or are probabilistically insignificant. Similarly, losses of all SW, and all DC are not discussed, but they would not lead to a blackout and direct core damage, due to design differences in support systems between the two plants. According to the information in the submittal, SUNPP-1 PRA did not consider common cause failures of support systems. Neither did it explicitly treat common cause spurious relief valve opening. In addition, failure of switchgear room cooling is stated in the SUNPP-1 PRA not to be an initiator, possibly due to long room heatup times.

(4) Containment bypass scenarios are not major contributors in either study. They are mentioned here because the CDF contribution is significantly higher in NVNPP-5 PRA than in the SUNPP-1 study. While not important in the Level 1 PRA, such differences may be important in Level 3 PRA analysis (that involves radiological releases from the plant).

The containment bypass contribution, which is much larger at NVNPP-5, might be a significant Level 3 concern. The difference in this contribution is due mainly to inclusion of certain pathways at NVNPP-5, which are not discussed in SUNPP-1, and it is not clear from the SUNPP-1 report why these pathways are not important (is it due to design differences or were they simply not considered because they are not important from a Level 1 perspective). Such pathways include RCS coolant bypass through the injection lines of the HPI and the LPI systems.

Other differences are less important, and are due to generally higher CCDF in NVNPP-5 due to design and data differences, and generally higher initiating event frequencies.

4. UNCERTAINTIES AND QUESTIONS TO BE RESOLVED

The explanations offered above for the differences in PRA results observable in Tables 1, 2 and 3 are subject to several uncertainties that cannot be resolved from the available PRA summary report documents. The following questions would be a start on the pathway to discern further causative factors:

- (1) What are the design and operational differences between NVNPP-5 and SUNPP-1 that lead to differences in assessment of the conditional probability of sump plugging?
- (2) Why is there an apparent difference in treatment of common cause support system failures and common cause spurious relief valve opening as initiating events?
- (3) Why is there a difference in importance of containment bypass scenarios and apparent difference in considered bypass pathways?
- (4) Does SUNPP-1 have the same LPI/RHR common mode failures as NVNPP-1 (due to common RHR line to the RCS and due to common electrical bus dependence that is also shared with secondary cooling systems)?
- (5) Additional description of the SUNPP-1 HRA treatment would help to provide a better comparison with the NVNPP-1 HRA treatment. Both reports would benefit from additional information in different areas.

5. CONCLUSIONS

The three methods used in the study, for explaining the calculated risk profile differences, sometimes overlap and sometimes complement each other, leading to causative factors of different levels of significance. Differences in LOCA and secondary breaks calculated risk profiles are due mainly to different assumptions or

limitations in the two PRA studies, while differences in transients (particularly support system initiators) are apparently due to design differences. Second order differences in design, data and assumptions all tend to increase the absolute value of CDF contributions at NVNPP-5 from various causes. Further information contained in the Russian versions of the PRA documents would be necessary in order to go deeper in the analysis than what has been presented here.

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